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EXECUTIVE SUMMARY

From June 10 through 21, 1991, the Office of Nuclear Reactor Regulation (NRR) conducted an inspection to assess the effectiveness of engineering and the design control process for modifications installed during the recently completed steam generator replacement project (SGRP) at Palisades. Region III requested that NRR perform this inspection to provide an independent assessment of the technical significance of regional inspection findings regarding analyses of piping and pipe supports.

The NRR inspection team validated the previously identified Region III concerns regarding weaknesses in the design control program related to inadequate control of licensing commitments for safety-related (seismic Category 1) piping systems, poorly documented or incomplete calculations for piping analyses, undocumented engineering judgements, and ineffective control of contracted engineering services in the piping analysis area.

The team found that inadequate control had been exercized over licensing commitments as evidenced by the fact that the licensee had modified the licensing basis for seismic Category 1 piping systems by making a change to the final safety analysis report to permit the permanent use of increased allowable stresses which had only been approved by the NRC for interim use. The team found that the licensee had employed the square root of the sum of the squares (SRSS) methodology for calculating the seismic input for a two-dimensional earthquake contrary to the commitment to apply the vertical and horizontal seismic loads simultaneously and combine them by the absolute sum. The team found that the licensee had not complied with American National Standards Institute (ANSI) standard B31.1 regarding (1) stress allowables for pipe supports under hydrostatic test conditions and (2) the required consideration of friction forces on pipe supports caused by thermal loads. The team also observed a number of other inconsistencies in the design specifications for the piping and pipe supports. The team concluded that the instances of improper translation of the licensing bases and commitments into design specifications indicated significant weaknesses in the design control program and in configuration management.

The team concluded that the number of concerns regarding the manner in which the main steam piping system had been analyzed jeopardized the adequacy of the calculational conclusions. The team felt additionally that the field interference, found by the licensee to exist between the main steam piping and the first pipe whip restraint inside containment during system hot functional testing, had not been appropriately modeled and that a number of engineering judgements had not been documented in the analyses. The team found a number of other errors and instances of undocumented engineering judgements in calculations and engineering analyses performed by the licensee and the licensee's contractor. The team considered the composite calculational errors to indicate an inattention to detail in the performance of the analyses and in the design review checking process. The team concluded that more rigorous licensee technical overview of engineering contractor work output was warranted.

The team also found that the licensee was unable to retrieve a number of original performed design calculations. Problems also existed with the control of revisions to drawings. The team concluded that these observations also indicated weak configuration management control by the licensee.

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1.0 INTRODUCTION

A U. S. Nuclear Regulatory Commission (NRC) Region III inspection (Inspection Report 50-255/90-25) identified a number of problems with engineering design and analysis work related to the design of piping and pipe supports for the recently completed steam generator replacement project (SGRP) at Palisades. The inspection raised specific concerns about the adequacy of the design verification measures, the use of questionable design assumptions, and the lack of adequate corrective actions. Region III requested that the NRC Office of Nuclear Reactor Regulation (NRR) perform an inspection to provide an independent assessment of the technical significance of the regional inspection findings regarding the analyses of piping and pipe supports.

In 1989 the NRC staff identified concerns regarding weaknesses in the Palisades design control program during both a regional engineering team inspection and a regional snubber reduction program inspection. During these inspections, the staff also found weaknesses related to undocumented engineering judgements, inadequate design verification, inadequate design control, drafting errors, and unauthorized design changes.

The NRC headquarters staff performed the current inspection at the request of the Region III office to provide an additional evaluation of the effectiveness of the licensee's design control program, provide an independent assessment of the significance of the Region III inspection findings in the piping analyses area, and judge the effectiveness of licensee corrective actions regarding design control to date.

The team reviewed engineering documentation related to plant modifications to assess conformance with Consumers Power Company (CPCo) design control requirements. The team focussed principally on design work completed at the same time as the design work examined in the recent Region III inspection. The team reviewed a representative population of piping and pipe support calculations that had been examined by the regional inspectors and reviewed other piping, pipe support, and mechanical systems modifications performed under the current licensee design control program.

The team characterized its findings in this report as deficiencies or observations. Deficiencies are errors, inconsistencies, apparent procedural violations or deviations regarding specific licensing commitments, specifications, procedures, codes, or regulations. Follow-up action consisting of resolution by the licensee and possible NRC evaluation or inspection are required. Observations address matters considered appropriate to call to the licensee's attention. Follow-up actions by the NRC, however, are not required.

2.0 REVIEW OF LICENSING BASIS AND PIPING AND PIPE SUPPORT DESIGN SPECIFICATIONS

During previous Region III inspections, a number of concerns were identified with the licensee's piping design specifications including the adequacy of the original seismic response spectra and concerns with the application of the original response spectra in current modification analysis. The team reviewed, for consistency, the seismic response criteria specified in the Palisades Final Safety Analysis Report (FSAR) at the time of issuance of the provisional operating license, hereafter referred to as the original FSAR design criteria, the current version of the FSAR, other docketed correspondence and the piping and pipe support design specifications. The results of the team's review are discussed below.

2.1 Review of Licensing Basis

The team reviewed the original seismic design criteria for Palisades as described in Appendix A of the FSAR and other docketed correspondence. In response to Inspection and Enforcement Bulletin (IEB) 79-14, "Seismic Analysis for As-Built Safety-Related Piping Systems," the licensee informed the NRC that some seismic Class 1 piping did not conform to the original FSAR acceptance criteria. The licensee provided an evaluation of the nonconformances and planned corrective actions which included the use of interim criteria to determine the operability of piping systems. These interim criteria utilized piping allowable stress criteria from the 1976 Winter Addenda of the 1974 Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Subsection NC. The licensee also committed to perform any modifications needed to upgrade the plant to the original FSAR criteria by the end of the next scheduled refueling outage. In a safety assessment of April 25, 1980, the NRC staff found the allowable stress criteria acceptable for interim use.

Subsequently, the licensee revised the FSAR to incorporate the interim stress allowables as permanent acceptance criteria for future analysis. The licensee reviewed the change under Section 50.59 of Title 10 of the Code of Federal Regulation (10 CFR 50.59) and informed the NRC of the change. The NRC staff has no safety evaluation that accepted the changes to the original licensing criteria. To assess the licensee's basis for changing the original FSAR licensing criteria, the team reviewed the minutes of a special meeting on August 22, 1980, of the Plant Review Committee (PRC) in which the PRC discussed the proposed change to the FSAR. The PRC concluded that the NRC safety assessment did not place any time limits on the acceptability of the interim criteria. The team considered that the PRC's interpretation of the NRC safety assessment was incorrect since the use of the criteria was explicitly authorized for one operating cycle. The team considered the licensee's extrapolation of the interim piping allowable stress values represented a reduction in design margin from the original licensing basis that should have been reviewed and approved by the NRC staff (see Appendix A, Deficiency D-1).

The team identified a second concern with the implementation of the original licensing basis criteria involving the combination of the vertical and horizontal earthquake response components in the piping analyses. In the Palisades FSAR, the licensee stated that the seismic design to the Palisades plant was based on simultaneously combining the responses of a horizontal component of the earthquake with the vertical component. This is commonly referred to as a two-dimensional or 2-D earthquake input which was used in the design of several older nuclear power plants. Palisades Technical Specification M-195(Q), "Requirements for the Design and Analysis of Palisades Plant Safety Related Piping and Instrument Tubing," Revision 1, May 9, 1990, specified that the horizontal and vertical components be combined by the square root of the sum of the squares (SRSS). The team considered that the horizontal and vertical earthquake components should have been combined by the absolute sum method since this has been the commonly accepted practice for plants using a 2-D earthquake input. The team considered the licensee's use of the SRSS combination for the 2-D earthquake did not meet the FSAR commitment to apply the loads simultaneously (see Appendix A, Deficiency D-2).

2.2 Review of Piping and Pipe Support Design Specifications

Palisades Technical Specification M-195(Q) contained the requirements for the design of piping, and Technical Specification C-173(Q)," Technical Requirements for the Analysis and Design of Safety Related Pipe Supports," Revision 2, November 21, 1990 contained the pipe support design requirements. The team reviewed these specifications for consistency with the design criteria in the Palisades licensing basis.

The team noted that both documents contained two sets of criteria, one set to be applied to existing designs and another set to be applied to new designs. The licensee had developed two sets of piping analysis criteria because the implementation of current seismic analysis techniques and the use of new seismic input developed as part of the NRC's Systematic Evaluation Program (SEP) were conditions for the NRC staff's acceptance of the licensee's use of ASME Code Case N-411 for new designs. The licensee also wanted to retain piping analysis criteria that were consistent with the original licensing basis for existing designs. The team took this into consideration in its review.

Technical Specification C-173(Q), Tables 1.0 and 2.0, contained loading combinations and allowable stresses for existing and new pipe support designs. Both tables contained separate sets of allowable stress limits applicable to either structural steel members designed to the American Institute of Steel Construction (AISC) Manual of Steel Construction or standard component catalog items purchased from pipe hanger manufacturers. The team found that the stress limits for the hydrostatic test loading specified for structural steel members were higher than the normal limits specified in the AISC Manual and did not comply with the requirements specified in the piping code of record, American National Standards Institute (ANSI) B31.1-1973 , (see Appendix A, Deficiency D-3). In addition, the licensee had not demonstrated that allowable stress limits met the piping code requirements for standard component catalog items.

Technical Specification C-173(Q), Section 5.4.2, specified criteria for evaluating the effects of friction loads on existing pipe supports. The criteria stated that friction loads shall be assumed to act simultaneously, with thermal and dead loads only. However, the criteria only required calculation of the friction force caused by dead load. These criteria did not comply with the requirements specified in the piping code of record, ANSI B31.1 (see Appendix A, Deficiency D-4). The team noted that the criteria specified for new supports in Section 6.3.2 of Technical Specification C-173(Q) considered both the dead load and thermal load in the calculation of the friction force.

The team had several additional comments on the adequacy of the design specifications for piping and pipe supports. Although the team did not characterize these comments as deficiencies, the team believed that they should be considered in resolving the licensing basis issues and in revising the technical specifications (see Appendix B, Observation O-1).

In particular, the team considered many of the distinctions made between existing and new design criteria in the specifications to be arbitrary and lacking in a sound technical basis. At the exit meeting held with the licensee, the team noted the following:

(1) The design specification for piping analysis, M-195(Q), stated that the missing mass correction did not need to be considered in the seismic analysis using the existing design criteria. The missing mass correction, sometimes referred to as the zero period acceleration (ZPA) load, occurs as a result of the response spectra analysis technique which does not consider higher frequency vibrational modes in the seismic analysis. The additional modes could be significant in certain cases.

(2) Attachment G of design specification for piping analysis, M-195(Q), contained guidelines for reconciling the as-built piping configuration to the as-designed. These guidelines are from Electrical Power Research Institute (EPRI) Report NP-5639, "Guidelines for Piping System Reconciliation (NCIG-05, Revision 1)," May 1988. The statement of scope for the document indicates the following "the document should be restricted to piping systems analyzed and qualified on the basis of currently accepted design criteria..." The use of this document may not be consistent with some of the technical positions for criteria specified in M-195(Q) for existing designs.

(3) The design specification for piping analysis, M-195(Q), allowed the use of the independent support motion (ISM) method of analysis. However, the ISM method of analysis is a more recent analytical method, and the design specification did not clarify which set of design provisions applied to its use.

(4) The design specification for pipe supports, C-173(Q) contained generic multiplication factors to increase the normal load capacity given by the vendor for standard component supports for the faulted load cases. These generic multiplication factors should be technically justified.

2.3 Summary

The inspection reviews identified cases where the licensing bases had not been properly translated into CPCo design specifications. This was evidenced by an the use of interim approved stress allowables on a permanent basis and the improper summation of earthquake components. The team additionally found cases where code requirements that were a part of the licensing basis had not been integrated into the design specification. This occurred with regard to definition of allowable stresses under hydrostatic loads and exclusion of thermal loads in the generation of friction forces. The team considered lack of rigor in the process of translating licensing bases and commitments into the design guides to be indicative of ineffective design control practices.

3.0 REVIEW OF PIPING AND PIPE SUPPORT CALCULATIONS

Previous Region III inspections had documented concerns regarding the adequacy and accuracy of design calculations for safety-related piping and pipe supports. The team reviewed a sample of the piping, pipe support and civil design calculations prepared to support the SGRP. The team reviewed these calculations to determine if they were technically correct, if they conformed to the governing procedures of the licensee and licensee's contractor, and if the pre-existing design basis was adequately addressed. The team reviewed calculations performed in support of facility change (FC) packages FC-911, FC-913, FC-893 and FC-894.

3.1 Main Steam Line Analysis

The SGRP included modifying and reanalyzing the main steam lines. The team reviewed the piping stress analysis and related calculations for pipe support and whip restraints for the main steam line from steam generator B to its containment penetration. The team found a number of nonconservative assumptions and methodologies in these analyses as discussed below.

The team found that the licensee had not addressed ZPA effects although the line included several long piping runs. In performing the piping analysis, the licensee had also combined the computed vertical seismic load with the larger of the two horizontal seismic loads by the SRSS method rather than by the absolute sum method. In addition, the team found that the analysis did not incorporate the evaluation of an anchor point movement at the location of the steam generator nozzle resulting from the vertical growth of the steam generator caused by sustained pressure (see Appendix A, Deficiency D-5).

During hot functional testing, the licensee had identified thermal binding of both main steam lines with a pipe whip restraint inside containment. To address the effect of thermal binding on the seismic response of the main steam line, the licensee performed a supplemental gap calculation that postulated impact at the location of thermal binding under a seismic event. The team had several concerns with the adequacy and completeness of the gap calculation such as the inconsistency between the actual piping geometry used in the analyses, incorrect consideration of seismic anchor movements, and incomplete analysis of thermal loads (see Appendix A, Deficiency D-6).

The main steam line between the containment penetration and the steam generator nozzle is supported by two constant supports and a pipe stanchion. The team identified a number of omissions in the calculations performed to qualify the supports. The team found that the swing angle checks for the constant supports were not correct or complete. The constant supports were installed on a pipe whip restraint, along with hanger rods. The team found the hanger rods were not analyzed for seismic loads. The calculation for the constant supports also did not include a deflection check to confirm that adequate travel existed in the constant supports during a seismic event. The pipe support calculation for the pipe stanchion did not address the revised and larger vertical thermal displacement that was documented in the piping analysis and that appeared to be larger than an "as-built" vertical clearance, which would subject the support to an unanalyzed thermal load. The team also confirmed two concerns identified by the regional inspection: (1) a deflection check in the calculation for the pipe stanchion was incorrect, and (2) a load had arbitrarily been distributed between the various components of the support rather than distributed on the basis of support stiffness (see Appendix A, Deficiency D-7).

The team identified problems with the calculation which was performed to requalify the whip restraint that was found to bind with the main steam line. The team found the calculation addressed the in-plane thermal loads that were documented in piping stress analysis, but did not consider the out-of-plane loads induced by friction. The measured pipe movement at the point of thermal binding from the cold to the hot condition appeared to exceed the design gap clearance originally specified for the whip restraint. Thus, the original design may not have adequately addressed the effects of the piping thermal growth. The team found that the licensee had performed checks using design data to confirm the gap clearances between the main steam legs and the remaining whip restraints rather than making field measurements. The checks showed inadequate clearances if the effects of piping radial thermal growth or lack of concentricity were considered (see Appendix A, Deficiency D-8).

During the inspection the team noted several observations as documented in Appendix B of this report. The team noted that the licensee could not retrieve the original calculations for the main steam containment penetration or the whip restraints for the main steam lines. Since these components appeared to be subject to large thermal binding loads that may not have been a part of the original design for these components, the team considered that the licensee should have reevaluated the original calculations (see Appendix B, Observation O-2).

The team also identified two span lengths for the vent lines from the main steam lines that are four to five times greater than the span lengths that the FSAR specified for field-routed small bore pipe. The team questioned if the licensee had consistently field routed small bore pipe in accordance with the FSAR criteria (see Appendix B, Observation O-3).

3.2 Sampling System Tubing Analysis

The SGRP included the rerouting and modification of several piping and tubing lines associated with the steam generator sampling system. Even though these systems consisted of small diameter piping and tubing, the licensee had analyzed them using rigorous computer techniques instead of the simplified techniques normally used for small diameter piping. The team selected three pipe stress analyses for review. The licensee had evaluated the two main steam sample tubing problems according to the criteria for existing systems and evaluated the steam generator water sampling line according to the criteria for new systems.

Although the licensee had rerouted portions of both steam sample tubing runs because of the change in elevation of the steam line nozzle connection points, the licensee analyzed both tubing runs using the original licensing basis criteria. The team noted that the tubing in the analysis problem spanned between the elevations of 645 feet and 667 feet 8 inches. However, the response spectra input used in the analyses only enveloped up to the elevation of 649 feet. This was the response spectra used in the original system design analyses.

This issue was previously identified in NRC inspection report 50-255/90-25. The team also noted that both calculations contained an assumption on the material properties of the short piece of existing piping from the main steam line branch connection to the tubing coupling. The calculations stated that this assumption did not need to be confirmed because of the short length of the existing piping and its proximity to an analytical anchor. The team verified the assumed material properties were correct by review of the piping class sheets in Specification M260. The team considered the original disposition as stated in the calculation to be technically inadequate (see Appendix A, Observation O-4).

Both tubing calculations contained thermal analyses of two separate line conditions. The first condition applied to the entire tubing run at the main steam temperature for steam flow through the line. The second condition applied to a thermal attenuation from the main steam temperature to the ambient temperature when there is no steam flow through the line. In evaluating the second condition, the licensee assumed that a relatively long length of the tubing was at main steam temperature. This assumption was based on an engineering judgement that did not have a documented basis. The team questioned if this assumption was conservative since the tubing near the main steam line was attached to the steam generator and the assumption of main steam temperature in the tubing would not result in significant differential thermal movement between the piping nozzle and the support attachments (see Appendix B, Observation O-5).

The licensee had analyzed the new water sampling line with the criteria used for new systems, which included the use of spectra input based on ASME Code Case N-411 damping. The piping and tubing in the analysis problem spanned between elevations 597 and 661 feet. The response spectra input enveloped the internal structure spectra up to 649 feet. The team found the input spectra used in this analysis did not meet the licensee's commitment for the use of ASME Code Case N-411 damping (see Appendix A, Deficiency D-9). This issue of spectra enveloping had been previously identified for the steam generator recirculation system during the Region III inspection.

3.3 Blowdown and Recirculation System Analysis

The SGRP included removing the 2-inch bottom blowdown piping and replacing it with 4-inch piping to increase the capacity of the blowdown system. Since the licensee considered this system was new, it analyzed the system using the criteria for new systems which included spectra input based on ASME Code Case N-411 damping. The team reviewed the piping stress calculation and several of the associated support calculations for the blowdown line.

During plant heatup the licensee found that the blowdown piping was not bearing on pipe support H-9. The team found the licensee had revised the pipe stress calculation to address the discrepancy by adding a statement that an additional deadweight load would be imposed on the containment penetration and that the penetration was adequate to accommodate the increased load. However, the team considered the evaluation incomplete in that it did not address the pipe support adjacent to support H-9 that would also have to support the increased deadload (see Appendix B, Observation O-6). The licensee subsequently evaluated the adjacent support and found that it also was adequate to accommodate the increased load.

Pipe support calculation SGBB-PD-H1, Revision 7, March 11, 1991 evaluated a deadweight hanger on the blowdown piping. This hanger was supported from a horizontal length of tube steel that was welded to the bottom flange of a structural I-beam tie strut. This configuration resulted in a local torsional load on the bottom flange of the tie strut. The tie strut was evaluated in civil calculation C-008, Revision 4, August 24, 1990. The team found that the evaluation of the tie strut included the evaluation of loads from the attached piping but did not address the local torsional load on the bottom flange of the I-beam (see Appendix A, Observation O-7). The licensee evaluated the local stresses and demonstrated that they were within acceptable limits.

The team reviewed two calculations that each qualified a pipe whip restraint design, and one calculation that qualified two modified blowdown penetrations and two modified recirculation penetrations. The calculations for the whip restraints appeared to be correct and complete. However, the team noted that calculation for the penetrations appeared to address a rupture moment incorrectly and did not adequately address a combination of rupture loads (see Appendix B, Observation O-8).

3.4 Summary

In summary, the team characterized most of its findings regarding the piping and piping analyses as observations since they appear to lack safety significance. However, the observations are evidence of inattention to detail in the performance of the analyses and the checking process. The team concluded that the number of concerns raised regarding the manner in which the main steam piping had been analyzed warranted additional attention to the documented details of that analysis. The team considered the existing main steam line analysis to be deficient because of the lack of clarity and completeness.

Further, the team concluded that the number and types of calculational errors demonstrated that the design control program was not being systematically implemented and that more rigorous licensee technical overview of contractor engineering work products was warranted.

4.0 REVIEW OF FACILITY CHANGE PACKAGES AND MECHANICAL DISCIPLINE CALCULATIONS

To obtain additional insights into the effectiveness of the design control process, the team reviewed five field change (FC) and two specification change (SC) packages. The team also reviewed 15 additional mechanical systems calculations prepared by the licensee's engineers and the licensee's contractor. Related documentation provided by the licensee's engineering staff in response to questions by the team was also reviewed.

4.1 Review of Facility Change Packages

In general, the team found the FC and SC packages to be technically adequate to support the modifications to the plant and to be consistent with the applicable administrative procedures: 9.03, "Facility Change," 9.03A, "Facility Change for SGRP," and 9.04, "Specification Changes." The appropriate engineering disciplines reviewed the change packages and provided comments on the design bases, codes and standards, correctness of the design and calculations, constructibility, testing, and the required revisions to documentation for the plant. The engineer responsible for the modification resolved the comments and made notations to that effect on the design review sheets. However, in many cases the team could not determine if the reviewer (or the reviewer's supervisor) agreed with the resolution of the comments. Some of the reviewers had indicated their concurrence with the resolution of their comments by signing the design review sheets, while others had not. Even though this practice was consistent with the administrative procedures, the team considered the lack of documented concurrence to be a weakness. A formal documented concurrence by the reviewer would help ensure that technically significant comments were not dismissed as inappropriate or incorrectly resolved because of a misinterpretation of the comment.

4.2 Review of Mechanical Discipline Calculations

The team reviewed the licensee's calculation index. The index identified each engineering analysis by the FC or SC number, system or component codes, title and other identifiers in accordance with administrative procedure 9.11, "Engineering Analysis." The team found that because the calculations were not identified by discipline, the calculations could not be easily identified and retrieved according to discipline. In addition, the procedures did not require that safety-related calculations be identified as such on the documents. To determine if a calculation was safety-related, the team had to review the calculation. The team found that the same procedures were used for preparing, reviewing and approving both safety-related and nonsafety-related calculations. The team considered the large number of engineering analyses produced each year might warrant a method for uniquely identifying the safety-related calculations to help focus engineering attention to them according to their importance to plant safety.

The team reviewed calculations from facility changes, specification changes, and the calculation index. The team verified the validity of the input data and assumptions, the applicability of the calculational methodology and the reasonableness of the results. Many of the team's specific comments on the calculations (see Appendix B, Observation O-9) again appeared to be characterized by inattention to detail in both the performance of the analysis and checking process.

The licensee obtained from the piping class summary sheets the design pressures and temperatures used in the calculations supporting piping modifications. The team reviewed the applicable design data contained in these sheets for the auxiliary feedwater (AFW) system. The licensee had revised the piping class summary sheets in 1985. The licensee could not retrieve the supporting documents or calculations for the revisions to the design ratings and service conditions for the AFW system piping. The service class description for AFW line EB-10 also appeared to be inconsistent with the piping diagram. The licensee could not retrieve the original design documents which supported the data in the piping class summary sheets. Therefore, the team could not determine if the installed system components meet the original or the revised design ratings. The team considered this lack of design basis documents and the inability to retrieve either the original calculations for the containment penetrations or the whip restraints (see Section 3.1 Observation O-2) to be evidence of weak configuration management.

4.3 Summary

While none of the concerns appeared safety significant, the team identified weaknesses in the mechanical system calculations that reflect less than fully adequate application of design control measures.

5.0 REVIEW OF CORRECTIVE ACTION PROGRAM

The team reviewed the portions of the licensee's corrective action program related to the maintenance and SGRP outages of 1990. This included reviewing corrective action documents and administrative procedures. During the SGRP outage, the Bechtel Power Corporation was the lead contractor and used several of its administrative procedures to process corrective action reports. The licensee had verified that these procedures were compatible with its own administrative procedures. The team reviewed specific corrective action documents including nonconformance reports (NCRs), field change requests (FCR's) field change notices (FCNs), and deviation reports (DRs).

5.1 Dispositioning of Nonconforming Conditions

The team reviewed a number of NCRs documented during the SGRP. The team found that the licensee had adequately dispositioned the issues presented in these NCRs. Significant issues identified on NCRs were resolved by licensee deviation reports. The team also reviewed a number of FCRs and FCNs associated with FC-893 and FC-895 most of which dealt with specific field installations that could not be completed as designed. These changes appeared to be dispositioned adequately by drawing revisions and reanalyses. The team reviewed the list of deviation reports for 1990 and 1991 and selected reports that appeared to concern piping or pipe support discrepancies. The team identified one deficiency and made one observation during the review of the DRs. The deficiency concerned delay of corrective action for a leaking weld.

At the end of the May 1990 maintenance outage, the licensee identified a leaking weld on the containment spray header inside containment. The licensee initiated a work order but performed no repairs before the completion of the outage. The licensee delayed the weld repair until the 1990 refueling outage apparently based on engineering judgement that the weld was structurally adequate to maintain header integrity. This judgement was not documented. However, paragraph IWA 5250 of Section XI ASME of the Code does not allow any through wall leakage in ASME Class piping. The piping in question was ASME Class 2 and was in the pressurized portion of the system. Therefore, the team concluded that the licensee should have repaired the weld defect before returning the plant to power operations (see Appendix A, Deficiency D-10).

While reviewing deviation reports, the team observed problems pertaining to the updating of controlled drawings. The team could not verify that the process for updating drawings was adequate to ensure that the controlled drawings were being revised promptly and that engineers were assured of using the most recent revisions to drawings for plant modifications.

The team found that administrative procedure 10.44, "Design Document Control and Distribution," did not state any specific process to update controlled drawings for specification changes or for changes resulting from the dispositioning of deviation reports as it did for facility changes. Section 9.5 of the procedure stated that the responsible engineer was to submit to the Document Control Center (DCC) an informational copy of the drawing being revised and Section 9.6 stated the plant drafter shall forward the Document Change Request (DCR) package to DDC. The DCC was to then stamp aperture cards as "currently being revised." DCC maintained files for each engineer and was to place copies of open DCRs in his or her file until the drawing was revised. The team found three instances in which (1) no DCR forms were maintained in the responsible engineers file folder for several drawings requiring update; (2) a marked up copy of the drawing was not with the control copy of the drawing; and (3) aperture cards were not stamped (see Appendix A, Observation O-10).

The DCC personnel acknowledged they had not completed work on a large number of drawing changes for the SGRP. The licensee stated this backlog resulted from the fact that the DDC was understaffed and that the design organization had recently moved from the corporate office to the site. The licensee is increasing the number of DDC personnel and expects the design organization's move to the site will eventually improve the timeliness of drawing revisions. However, the team remained concerned that drawings that were being revised did not appear to be documented on DCRs for the DCC and that controlled drawings were not annotated to indicate that revisions were in progress to ensure that engineers designing subsequent modifications use the most recent drawings.

5.2 Safety-Related Piping Reverification Program

Responding to previous NRC inspection findings concerning the adequacy of the licensee's response to IEB 79-14, "Seismic Analysis for As-Built Safety-Related Piping Systems," the licensee initiated a safety-related piping reverification program (SRPRP) to reverify the seismic adequacy of the design of safety-related piping of a diameter 2 1/2 inches and larger.

This program was divided into two phases. In Phase I the licensee performed system walkdown inspections and reconciled discrepancies between the design and as-built configuration for the 18 stress packages associated with eight systems that are required to maintain primary coolant system inventory levels, boron concentration, and emergency cooling capabilities. The licensee had a third party independent review of selected walkdown inspections and analyses. The third party identified at least one condition that did not meet the FSAR design basis. The team considered the use of third party reviewers to be a program strength.

The team found that 8 of 18 stress packages were complete and complied with FSAR requirements, 6 packages met interim operability analyses, and the licensee had completed walkdown inspections of the remaining 4 packages. The licensee had recently hired a new contractor to complete the remaining portion of Phase I. Results from Phase I prompted the licensee to make several hardware modifications. The licensee will evaluate the data accumulated during Phase I for determining the scope of Phase II.

The team reviewed two of the walkdown stress packages for which reconciliation had not been completed. The team did not perform any walkdown inspections. The packages reviewed included portions of auxiliary feedwater (AFW) system and the discharge piping for the high pressure safety injection (HPSI) pump. The walk-down inspections were detailed and identified several good findings. The licensee had addressed physical discrepancies by issuing a work order to repair the discrepancy or by performing an engineering analysis to verify operability. In all but one case, the dispositioning of these discrepancies appeared adequate. The team found for hanger support DC1-H207, an unintentional restraint was identified and dispositioned by the licensee incorrectly by stating that the thermal movement of the hanger would be in the downward direction when it was actually a minimal amount in the upward direction.

5.3 Summary

The team identified that the licensee had generally effective corrective action programs. One outstanding exception was the erroneous disposition to accept through wall leakage of an ASME pipe.

The team found that although the licensee had successfully identified a number of discrepancies in the as-built piping configuration at Palisades, the team did not consider that the licensee could successfully reconcile the as-built plant configuration to the plant design basis until the licensee resolved the team's other concerns and observations regarding the FSAR design criteria, the piping specifications, the calculations and the drawing revision process weaknesses.

6.0 EXIT

On June 21, 1991 the team conducted an exit meeting at the Palisades site. Appendix C lists the representatives of CPCo and the NRC attending the exit meeting. During the exit meeting, the team summarized the scope and findings of the inspection.

At the exit meeting, Mr. D. P. Hoffman CPCo Vice President for Nuclear Operations, made the following commitments that were subsequently documented in a letter to the NRC of July 9, 1991:

- Submit for NRC staff review a revision to the FSAR which clarifies piping and pipe support design criteria;
- Upgrade the piping and piping design specifications to reflect the revised FSAR criteria;
- Develop a document to relate specification and procedural requirements;
- Suspend the implementation of piping and pipe support modifications until revision to the FSAR and specifications were complete;
- Continue with third party review of piping and pipe support analysis; and
- Complete an assessment of pipe and pipe support engineering to identify design engineering strengths and weaknesses, the root causes of weaknesses and suggested improvements to address the root causes.

Furthermore, the licensee has committed to reanalyze the main steam line design after revising the FSAR and design specifications.

APPENDIX A

Summary of Deficiencies

DEFICIENCY D-1

FINDING TITLE: Piping stress allowable limits

BACKGROUND:

The licensee described the original Palisades seismic design criteria in Appendix A of the FSAR (Reference 1). In response to Inspection and Enforcement Bulletin (IEB) 79-14 (Reference 2), the Consumers Power Company (CPCo) informed the NRC that some of the seismic Class 1 piping did not conform to the Palisades FSAR acceptance criteria. CPCo provided an evaluation of the nonconformances and the planned corrective actions in subsequent submittals to the NRC (References 3, 4, 5, 6). The corrective actions included the use of interim allowable stress criteria to determine the operability of piping systems. These interim criteria utilized piping allowable stress criteria from Reference 7. In Reference 3, CPCo committed to perform any modifications needed to upgrade the plant to the FSAR criteria by the end of the Cycle 4 refueling outage scheduled for 1981. The NRC performed a safety assessment and found the allowable stress criteria acceptable for interim use (Reference 8).

In a subsequent letter to the NRC (Reference 9) CPCo stated that it had revised certain pages of the Palisades FSAR in response to the IEB 79-14 work and that the revised FSAR page changes would be transmitted in a separate letter (Reference 10).

DESCRIPTION OF CONDITION:

The seismic design criteria for nuclear power plant piping systems have changed significantly since older plants, such as Palisades, were licensed. Older nuclear power plants generally were designed using less conservative seismic inputs and less rigorous analysis procedures than those used on the more current plants. These less conservative inputs and procedures were generally used with acceptance criteria for allowable stresses in piping and piping supports that are more conservative than those used on more current plants. In recognition of this, the NRC, in Revision 1 of IEB 79-14, requested that nonconformances be evaluated to either FSAR or other NRC approved acceptance criteria. Consistent with other staff positions taken during the implementation of IEB 79-14, the NRC staff allowed CPCo to use the higher allowable stresses specified in the ASME Code as interim criteria until the original FSAR design margins were restored.

In inspection report 50-255/90-25, the staff identified a number of concerns with the seismic input used to evaluate piping systems at Palisades. This included concerns regarding the adequacy of the original floor response spectra and the application of the original floor response spectra to current modification design analysis. CPCo's preliminary response to these concerns was that the original response spectra were considered adequate when Palisades was licensed and that the concerns raised were beyond the original licensing basis.

However, CPCo had changed the original licensing basis in the FSAR amendment submitted in Reference 10. CPCo could not locate an NRC staff safety evaluation that accepted this change to the original licensing basis. The pipe stress allowables used by CPCo did not conform to the original Palisades licensing basis. The NRC staff had apparently only reviewed and accepted these allowables for use as interim criteria. The team concluded that CPCo's change to the original FSAR licensing basis should not have been performed under 10 CFR 50.59 but should be reviewed and accepted by NRC staff.

REFERENCES:

1. Consumers Power Company Palisades Plant Final Safety Analysis Report, Amendment 14 (July 22, 1969) through Amendment 17 (November 14, 1969)

2. NRC Office of Inspection and Enforcement Bulletin (IEB) 79-14, "Seismic Analysis for As-Built Safety-Related Piping Systems," July 2, 1979; Revision 1, July 19, 1979; Supplement 1, August 15, 1979; Supplement 2, September 7, 1979.

3. Consumers Power Company letter to the NRC, February 14, 1980.

4. Consumers Power Company letter to the NRC, February 27, 1980.

5. Consumers Power Company letter to the NRC, March 11, 1980.

6. Consumers Power Company letter to the NRC. April 14, 1980.

7. 1976 Winter Addenda of the 1974 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NC.

8. NRC letter to Consumers Power Company, April 25, 1980.

9. Consumers Power Company letter to the NRC, September 26, 1980.

10. Consumers Power Company letter to the NRC, October 24, 1980.

DEFICIENCY D-2

FINDING TITLE: Combination of horizontal and vertical seismic inputs

BACKGROUND:

Appendix A of the FSAR (Reference 1) described the original Palisades seismic design criteria. The design criteria specified both vertical and horizontal seismic input. As discussed in the FSAR, the Palisades plant was designed based on combining simultaneously the horizontal component of the earthquake with the vertical component. This is commonly referred to as a two dimensional (2-D) earthquake input that was used in the design of several older nuclear power plants. Palisades Technical Specification M-195(Q) Section 5.10.4.1.2 (Reference 2) specified that the horizontal and vertical components be combined by the square root of the sum of the squares (SRSS) method.

DESCRIPTION OF CONDITION:

Regulatory Guide 1.92 (Reference 3) provides NRC staff's current guidance for combining earthquake components. This guidance allows the use of SRSS to com- [illegible] all three components of the earthquake motion. The common practice for [illegible] plants that used the 2-D earthquake input was to use the absolute sum method for the two earthquake components. The team considers the FSAR statement that the horizontal and vertical components were applied simultaneously to mean the absolute sum of those components was applied, not the SRSS. The team also considered the use of SRSS for a 2-D earthquake to be inconsistent with the industry's common practice.

REFERENCES:

1. Consumers Power Company Palisades Plant Final Safety Analysis Report, Amendment 14 (July 22, 1969) through Amendment 17 (November 14, 1969).

2. Palisades Technical Specification M-195(Q), "Requirements for the Design and Analysis of Palisades Plant Safety Related Piping and Instrument Tubing," Revision 1, May 9, 1990.

3. NRC Regulatory Guide 1.92, "Combining Model Responses and Spatial Components in Seismic Response Analysis" Revision 1, February 1976.

DEFICIENCY D-3

FINDING TITLE: Pipe support allowable stress limits during hydrostatic tests

BACKGROUND:

Palisades Technical Specification C-173(Q) (Reference 1) contains the design criteria for pipe supports. Tables 1.0 and 2.0 specify load combinations and allowable stress limits for existing and new pipe support designs. Section 4.1.1 of Technical Specification M-195(Q) (Reference 2) states that the original piping design code of record is USAS B31.1.0, 1967 Edition (Reference 3) and that the current piping analysis shall be done to ANSI B31.1, 1973 Edition (Reference 4) up to and including the Summer 1973 Addenda. The piping code contains criteria for the design of supports.

DESCRIPTION OF CONDITION:

Tables 1.0 and 2.0 of Technical Specification C-173(Q) specify an allowable stress limit for the design of the structure components of the lesser of 0.8 times the material yield stress or 1.3 times the normal AISC (Reference 5) allowable stress limit for the hydrostatic test load combination. Paragraph 120.2.4 in both editions of the B31.1 code does not allow an increase in the normal AISC allowable stresses for supplementary structrual steel for the hydrostatic load case. Paragraph 121.1.2 of the B31.1 code allows an increase in allowable stresses for standard component support items (catalog items), up to 0.8 times the material yield stress for the hydrostatic test load case. Tables 1.0 and 2.0 of Technical Specification C-173(Q) specify generic multiplication factors to increase the normal allowable stresses. CPCo needs to demonstrate that the use of the generic multiplication factors meet the piping code requirements.

REFERENCES:

1. Palisades Technical Specification C-173(Q), "Technical Requirements for the Analysis and Design of Safety Related Pipe Supports," Revision 2, November 21, 1990.

2. Palisades Technical Specification M-195(Q), "Requirements for the Design and Analysis of Palisades Plant Safety Related Piping and Instrument Tubing," Revision 1, May 9, 1990.

3. U.S.A. Standard Code for Pressure Piping, USAS B31.1.0-1967, "Power Piping."

4. American National Standards Institute Code for Pressure Piping, ANSI B31.1-1973, "Power Piping."

5. American Institute of Steel Construction (AISC) Manual of Steel Construction, Seventh Edition.

DEFICIENCY D-4

FINDING TITLE: Pipe support friction loads

BACKGROUND:

Palisades Technical Specification C-173(Q) (Reference 1) contains design criteria for pipe supports. Section 5.4.2 specifies the criteria for evaluating the effects of friction forces on existing pipe supports. The criteria state that friction forces shall be assumed to act simultaneously with thermal and dead loads only. However, the criteria for existing supports only require the evaluation of the friction force caused by dead load. The criteria for new supports, specified in Section 6.3.2 of the specification, requires that both the dead load and the thermal load be considered.

DESCRIPTION OF CONDITION:

During piping system heatups and cooldowns, the piping system expands and contracts causing the piping to slide past some of the rigid pipe support contact surfaces. Friction forces result from the bearing load imposed on the contact surfaces as the pipe slides past. These friction forces are in the direction of the pipe thermal movement and the magnitude of the friction force depends on the total load on the support as the pipe slides past. Paragraph 121.2.1 of References 2 and 3 requires that fixed pipe restraints and guides be structurally suitable to withstand the thrusts, movements, and other loads imposed by the thermal expansion and contraction of piping. Since thermal loads are imposed on the fixed supports during the expansion or contraction of the piping system, these loads should be considered in the evaluation of friction force.

REFERENCES:

1. Palisades Technical Specification C-173(Q), "Technical Requirements for the Analysis and Design of Safety Related Pipe Supports," Revision 2, November 21, 1990.

2. U.S.A. Standard Code for Pressure Piping, USAS B31.1.0-1967, "Power Piping."

3. American National Standards Institute Code for Pressure Piping, ANSI B31.1-1973, "Power Piping."

DEFICIENCY D-5

FINDING TITLE: Main steam line piping stress analysis

BACKGROUND:

A piping stress analysis (Reference 1) qualified the 36-inch diameter main steam line from the steam generator B nozzle to containment penetration 3. The piping was shown on the stress isometric drawing (Reference 2). The vertical leg of the piping run passes through a series of pipe whip restraints that were shown on the civil drawing (Reference 3).

DESCRIPTION OF CONDITION:

The team reviewed the piping stress analysis (Reference 1) and identified the following concerns:

(1) The piping stress analysis did not compute seismic axial loads caused by zero period acceleration (ZPA). This is also referred to as a missing mass correction. Subsection 4.4.2.4.1 of the Reference 4 specification does not require the effect of ZPA to be considered when a system is analyzed to the original plant design criteria which uses 0.5 percent damping. However, because the main system line consists of several long runs of piping (two horizontal legs of 37 feet and a vertical leg of 63 feet), consideration of the effect of seismic axial loads caused by ZPA appears warranted. This concern is further elaborated upon in Observation O-1 under the discussion of the "Missing Mass Correction."

(2) The piping stress analysis combined the computed vertical seismic load with the larger of the two horizontal seismic loads by SRSS rather than by the absolute sum. This method was in accordance with the method prescribed in Subsection 4.4.2.4.1 of the Reference 4 specification but is not in accordance with the requirements of Palisades FSAR Subsection 5.7.4 (Reference 5). This concern is further elaborated upon in Deficiency D-2.

(3) The Reference 6 calculation computed the thermal and pressure growth of the replacement steam generators for use in designing the thermal insulation for the steam generator. According to the Reference 5 calculation, the steam generator grows vertically because of sustained internal pressure by about 0.3 inch, which is four to five times greater than the threshold of 0.0625 inch for consideration of anchor point movements in Subsection 4.4.2.4.2 of the Reference 4 specification. However, the piping stress analysis did not address anchor point movement caused by sustained pressure. If considered, this would increase the magnitude of the vertical anchor point movement (because of thermal growth) that was used as input to the piping analysis at the location of the steam generator nozzle by about 15 percent.

REFERENCES:

1. Bechtel Calculation SGRP-PDS-034, "Pipe Stress Analysis of Steam Generator E50B Main Steam System Inside Containment," Revision 5, March 28, 1991.

2. Bechtel Drawing 03374 Sheet 1 of 1 (Q), "Stress Isometric, Main Steam System," Revision No. 1, December 5, 1988.

3. Bechtel Drawing C-163, "Containment Steam Line Rupture Supports," Revision 5, March 24, 1972.

4. Bechtel Plant Design Criteria Document 20557-G-001P, Revision 4, January 21, 1991.

5. Consumers Power Company Palisades Plant Final Safety Analysis Report, Amendment 14 (July 22, 1969) through Admendment 17 (November 14, 1969).

6. Bechtel Calculation SGRP-M-003, "SGRP-Palisades Nuclear Station Steam Generator Pressure and Thermal Growth," Revision 0, May 25, 1990.

DEFICIENCY D-6

FINDING TITLE: Main steam line gap analysis

BACKGROUND:

To address the effect of thermal binding between the main steam piping and the first pipe whip restraint inside containment on the seismic response of the main steam piping system, the licensee prepared a supplemental gap analysis based on an ASME paper methodology as part of the Reference 1 calculation. The gap analysis computed additional loads on the piping system caused by impact between the piping and the whip restraint during an earthquake.

DESCRIPTION OF CONDITION:

The team reviewed the gap analysis and identified the following concerns:

(1) The licensee assumed that a seismic anchor movement (SAM) of the pipe with respect to the whip restraint of 0.2 inch. However, Subsection 4.4.2.4.2 of the Reference 2 specification states that the containment shell and the containment internal structure SAMs occur in phase, and are approximately the same value for a given elevation. Since the main steam line is anchored to the containment liner at elevation 618 feet and the whip restraint is supported by the containment internal structure at elevation 623 feet 6 inches, the team considered the assumption regarding the occurrence of a differential SAM between the pipe and the whip restraint to be unjustified. Further, the thermal analysis, X6040, which was documented in the Reference 1 calculation, computed high thermal loads at the point of binding. The team judged if a second seismic analysis were performed using the same restraint geometry that was used in the thermal analysis, X6040, and a comparison of the analyses were made, the thermal loads would be shown to envelope the seismic loads indicating that the main steam line would maintain bearing contact with the whip restraint during an earthquake. Therefore, the team concluded that the assumption that impact occurs between the piping and the whip restraint did not have an adequate basis.

(2) The gap analysis assumed the piping geometry to be a straight run of simply-supported pipe. This geometry varied considerably from the piping geometry shown on the Reference 3 drawing.

(3) The Reference 3 drawing documented an assumed plan dimension for the piping between the containment penetration and the whip restraint. This dimension was critical in determining the magnitude of the equal and opposite thermal reactions imposed on the containment penetration and the whip restraint because they are directly proportional to the length of the pipe. The team reviewed piping fabrication drawings and concluded that the assumed dimension was shorter than the actual length of pipe subject to axial thermal expansion. Thus, the computed thermal loads would be smaller than the actual loads.

(4) Thermal analysis X6040, superseded the thermal portion of the original piping stress analysis X2003A, in the Reference 1 calculation. However, the licensee did not update the load summary sheet that was originally prepared for the containment penetration to incorporate the revised and substantially higher thermal loads from the new thermal analysis.

(5) The gap analysis did not compute any transverse displacements. These displacements should have been combined with the seismic displacements from the original stress analysis X2003A. The licensee should have used these "inertial plus impact" seismic displacements to recheck the swing angles documented in the Reference 5 and 6 calculations for constant supports H14 and H15, and to verify the gap clearances between the vertical leg of the main steam piping and the remaining whip restraints documented in the Reference 7 calculation.

REFERENCES:

1. Bechtel Calculation SGRP-PDS-034, "Pipe Stress Analysis of Steam Generator E50B main Steam System Inside Containment," Revision 5, March 28, 1991.

2. Bechtel Plant Design Criteria Document 20557-G-001P, Revision 4, January 21, 1991.

3. Bechtel Drawing 03374 Sheet 1 of 1 (Q), "Stress Isometric, Main Steam System," Revision 1, December 5, 1966.

4. Bechtel Drawing C-163, "Containment Steam Line Rupture Supports," Revision 5, dated March 24, 1972.

5. Bechtel Calculation MSB-PD-EB1-H14, "Pipe Support Design for Main Steam System, Steam Generator E50B, Support No. EB1-H14," Revision 6, February 21, 1991.

6. Bechtel Calculation MSB-PD-EB1-H15, "Pipe Support Design for Main Steam System, Steam Generator E50B, Support No. EB1-H15," Revision 5, April 4, 1991.

7. Bechtel Calculation C-044, "Structural Adequacy Evaluation for Main Steam Line Pipe Support Loads," Revision 3, February 13, 1991.

DEFICIENCY D-7

FINDING TITLE: Main steam line pipe supports

BACKGROUND:

Main steam line leg B from the steam generator nozzle to containment penetration 3 is supported at three locations:

1. By constant support hanger H14 at elevation 681 feet 5 inches.
2. By constant support hanger H15 at elevation 664 feet.
3. By pipe stanchion H16 at elevation 618 feet.

Hanger H15 is supported from the pipe whip restraint at elevation 673 feet 6 inches. References 1-3 document the pipe support calculations for these supports.

DESCRIPTION OF CONDITION:

The team a reviewed the calculations and had the following findings:

(1) The swing angle checks for supports H14 and H15 used the faulted displacements from computer generated analysis X2003A in the Reference 4 calculation which included the transverse displacements caused by the combination of the dead load, thermal load and safe shutdown earthquake. However, these checks did not address the following design elements:

   a. The larger thermal displacements computed in run X6040 which considers thermal restraint caused by binding and superseded the thermal portion of analysis X2003A.

   b. Any additional seismic displacements that the gap analysis documented in the Reference 4 calculation might have predicted. However, the gap analysis did not calculate these displacements.

   c. The ±0.5 degree construction tolerance that Reference 5 specifies to be used for swing angle checks when design rather than measured displacements are used to perform the check. This check is performed to confirm that the support will not bind under the imposed lateral displacements, and to compute the secondary loads that the pipe support must absorb.

(2) A rod hanger was installed on the pipe whip restraint to absorb the dead load of the constant support since the pipe whip restraint bracing apparently was designed only to resist the out-of-plane dead load of the restraint. The Reference 2 calculation explicitly qualified the rod hanger for the hanger dead load but did not address the faulted load condition which would require the rod hanger to absorb an additional seismic tensile load caused by the out-of-plane excitation of the restraint itself. The rod hanger did not appear to be able to absorb any additional load. However, the Reference 5 civil calculation which qualified the whip restraint for the imposed hanger loads explicitly deleted the rod hanger from the structural model that was documented in the calculation.

The civil calculation demonstrated that the whip restraint bracing could resist the faulted condition out-of-plane dead and seismic loads.

(3) The Reference 2 calculation for the pipe support did not document a deflection check. Subsection 5.7.1 of the Reference 6 design specification specified a default maximum allowable deflection of 0.0625 inch for pipe supports in the direction of the primary load. However, Subsection 5.7.4 of the Reference 6 specification noted that deflection of pipe whip restraints need not be considered. This assumption appears unconservative.

(4) The team identified the following concern for support H16, which is a pipe stanchion that rests on a concrete base. A steel frame constructed on the concrete base was designed for an "as-built" vertical clearance of 3/32 inches between the frame and the tubular stubs that are welded to the pipe stanchion, to prevent the support from moving upward. The thermal analysis X6040 documented in the Reference 4 calculation predicted a vertical thermal displacement of 0.157 inches for the pipe stanchion which is greater than the "as-built" vertical gap. Therefore, the stanchion stubs and steel frame appeared to be subject to an unanalyzed thermal load. However, the licensee indicated that this uplift did not actually occur and would revise the Reference 3 calculation to document this fact.

REFERENCES:

1. Bechtel Calculation MSB-PD-EB1-H14, "Pipe Support Design for Main Steam System, Steam Generator E50B, Support No. EB1-H14," Revision 6, February 21, 1991.

2. Bechtel Calculation MSB-PD-EB1-H15, "Pipe Support Design for Main Steam System, Steam Generator E50B, Support No. EB1-H15," Revision 5, April 4, 1991.

3. Bechtel Calculation MSB-PD-EB1-H16, "Pipe Support Design for Main Steam System, Steam Generator E50B, Support No. EB1-H16," Revision 5, April 3, 1991.

4. Bechtel Calculation SGRP-PDS-034, "Pipe Stress Analysis of Steam Generator E50B Main Steam System Inside Containment," Revision 5, March 28, 1991.

5. Bechtel Calculation C-044, "Structural Adequacy Evaluation for Main Steam Line Pipe Support Loads," Revision 3, February 13, 1991.

6. Palisades Technical Specification C-173(Q), "Technical Requirements for the Analysis and Design of Safety Related Pipe Supports," Revision 2, November 21, 1990.

DEFICIENCY D-8

FINDING TITLE: Main steam line pipe whip restraints

BACKGROUND:

The vertical portion of the main steam lines are restrained against postulated pipe rupture by the pipe whip restraints shown on the Reference 1 drawing. The containment internal structure supports the pipe whip restraints at elevations 623 feet 6 inches and 637 feet. The containment liner supports the pipe whip restraints at elevations 651 feet 6 inches, 662 feet 6 inches and 673 feet 6 inches.

DESCRIPTION OF CONDITION:

The team identified the following findings after reviewing the Reference 1 drawing and the calculations in References 2 and 4:

(1) The thermal binding of the main steam lines subjected the pipe whip restraint at elevation 623 feet 6 inches to high in-plane thermal reactions. Reference 3 and 4 document the thermal loads for the main steam lines. The Reference 4 analyses documented an upward (out-of-plane) thermal displacement of 0.405 inch for the main steam line at the point of thermal binding. As a consequence, the pipe whip restraint was subject to additional secondary loads caused by friction. The Reference 2 calculation analyzed the whip restraint for the faulted load condition (dead plus seismic plus thermal) but did not address the normal load condition (dead plus thermal plus friction).

The Reference 5 design specification governing the design of pipe whip restraints does not address friction, since thermal binding should not normally occur. However, the Reference 6 design specification governing the design of pipe supports addresses the effects of friction and documents standard methods of addressing friction either by applying the a friction force (the friction coefficient times the same of the sustained dead and the thermal bearing load) or by applying the axial thermal piping movement to the restraint. Applying the friction load to the whip restraint would impose an out-of-plane load of over 100 kips.

(2) The actual movement between the B main steam line and the pipe whip restraint is a minimum of 9/16-inch caused by thermal movement alone. The conclusion is based on the measured hot and cold gaps between the B main steam line and the pipe whip restraint at elevation 623 feet 6 inches that are documented in the Reference 4 calculation. The Reference 1 drawing specified a shimmed clearance of 9/16 inch (+0, -1/16 inch) for a minimum design clearance of 1/2 inch. This design gap should have provided sufficient clearance to accommodate seismic movement in addition to thermal movement. Therefore the original design may not have adequately addressed the effects of piping thermal growth.

(3) Reference 2 calculation checked the clearances between main steam lines and the pipe whip restraints at elevations 637 feet, 651 feet 6 inches, 662 feet 6 inches and 673 feet 6 inches. However, the Reference 2 calculation checked the faulted thermal and seismic displacements documented in the Reference 3 and 4 calculations against the design clearances specified on the Reference 1 drawing rather than with measured data. The Reference 2 calculation assumed that the main steam lines were concentrically aligned with respect to the restraints, and did not consider the 0.062 inch radial thermal growth of the pipe that was documented in the Reference 4 calculation. Several of the computed clearances were marginal. For example, the minimum computed clearances for the whip restraint at elevation 637 feet (immediately above the restraint where thermal binding occurs) were 0.074 inch for main steam line leg A and 0.039 inch for main steam line leg B. If an additional 0.062 inch were deducted to account for piping radial thermal growth, the adjusted clearances become 0.012 inch and 0.0 inch with bearing contact. The main steam lines were apparently not concentrically aligned, as noted by the disparity between the measured and design cold gaps for the restraint at elevation 623 feet 6 inches (0.239 inch, 0.648 inches measured, 0.5 inch minimum design). As a consequence, the Reference 2 calculation did not clearly confirm that adequate gap clearances existed between the main steam lines and the remaining whip restraints. The Reference 3 and 4 piping stress analyses did not consider the possibility of interference between the main steam line legs and the remaining whip restraints.

REFERENCES:

1. Bechtel Drawing C-163, "Containment Steam Line Rupture Supports," Revision 5, March 24, 1972.

2. Bechtel Calculation C-044, "Structural Adequacy Evaluation for Main Steam Line Pipe Support Loads," Revision 3, February 13, 1991.

3. Bechtel Calculation SGRP-PDS-033, "Pipe Stress Analysis of Steam Generator E50A Main Steam System Inside Containment," Revision 5, March 28, 1991.

4. Bechtel Calculation SGRP-PDS-034, "Pipe Stress Analysis of Steam Generator E50B Main Steam System Inside Containment," Revision 5, March 28, 1991.

5. Bechtel Civil Design Criteria Document 20557-G-001C, Revision 1, December 11, 1989.

6. Bechtel Plant Design Criteria Document 20557-G-001P, Revision 4, January 21, 1991.

DEFICIENCY D-9

FINDING TITLE: Response spectra input

BACKGROUND:

Facility change package FC-913, added a new water sampling line during the steam generator replacement project. The licensee analyzed this new sampling line using the criteria for new systems specified in Palisades Technical Specification M-195(Q) (Reference 1). The analysis of the new line was contained in calculation SGRP-PDS-020 (Reference 2). The calculation used the seismic spectra input developed based on ASME Code Case N-411 damping. The piping and tubing in the analysis problem spanned between elevations 597 feet to 661 feet. The response input used in the analysis enveloped the internal structure spectra up to an elevation of 649 feet.

DESCRIPTION OF CONDITION:

The NRC staff conditionally endorsed the use of ASME Code Case N-411 in Regulatory Guide 1.84 (Reference 3). CPCo agreed to meet the conditions in the regulatory guide and documented this commitment in Section 5.7.1 of Revision 3 of the Palisades Updated Safety Analysis Report.

Regulatory Guide 1.84 includes the condition that the damping values may be used only in those analysis in which current seismic spectra and procedures have been employed. In Section 3.9.2 of the Standard Review Plan (Reference 4) the NRC staff stated that an acceptable approach for the seismic analysis of equipment supported at two or more locations is to use an upper bound envelope of the spectra at all support attachment points. The NRC staff has required the use of enveloped response spectra as a condition for the use of ASME Code Case N-411 damping. The spectra input used for the water sampling line was not an upper bound envelope of all support attachment points.

REFERENCES:

1. Palisades Technical Specification M-195(Q), "Requirements for the Design Analysis of Palisades Plant Safety Related Piping and Instrument Tubing," Revision 1, May 9, 1990.

2. Bechtel Calculation SGRP-PDS-020, "Piping/Tubing Stress Analysis for 1" Water Sampling Line," Revision 3, February 21, 1991.

3. USNRC Regulatory Guide 1.84, "Design and Fabrication Code Case Acceptability - ASME Section III Division 1," Revision 24, June 1986.

4. NUREG-0800, "Standard Review Plan," Revision 1, July 1981.

DEFICIENCY D-10

FINDING TITLE: Leak from weld on containment spray header

DESCRIPTION OF CONDITION:

At the end of the 1990 maintenance outage, the licensee identified a leaking weld on the containment spray header inside containment. The licensee initiated a work order (Reference 1), however, the weld repair was delayed until the 1990 refueling outage. The licensee believed the delay to be acceptable based on judgement that the weld was structurally adequate to maintain the integrity of the header. The licensee did not however document the basis of this judgment. ASME Section XI Code (Reference 2) does not allow any through wall leakage in ASME Class piping. The piping in question is ASME Class 2 and is in the pressurized portion of the system. Therefore, the licensee should have repaired the weld defect before returning the reactor to power.

The licensee interpreted Reference 2 to allow repairs to be delayed by engineering judgement. After the licensee identified the leak, ASME published an interpretation of the Code (Reference 3) reiterating the conclusion that engineering judgement permitting through wall leakage was unacceptable and that repairs were required. This interpretation prompted the licensee to initiate a deviation report (Reference 4) to document the fact that it had not acted in accordance with the ASME Code requirements. The team considered this to be a failure to properly evaluate ASME Code requirements and a failure to implement prompt corrective action for a known defect as required by Reference 5.

REFERENCES:

1. Palisades Work Order 24003211 May 7, 1990.
2. ASME Section XI 1983 Edition, IWA 5250.
3. ASME Section XI Code Interpretation XI-1-89-40, May 24, 1990.
4. Deviation report, D-PAL-90-313, November 26, 1990.
5. Section XVI, "Corrective Action", of Appendix B to Part 50 of Title 10 of the code of Federal Regulations.

APPENDIX B

Summary of Observations

OBSERVATION O-1

OBSERVATION TITLE: Comments on piping and pipe support specifications

Palisades Technical Specification M-195(Q) (Reference 1) provides criteria for the design and analysis of safety-related piping and instrument tubing. The specification defines criteria that apply to existing systems analyzed to the original plant design methods and criteria that apply to newly designed systems analyzed with current seismic input and the damping values specified in ASME Code Case N-411. The team reviewed the piping specification for conformance with licensing commitments and for the consistent application of good industry practice. The team made the following observations.

(1) Section 5.5.3, "Support Mass," states that support mass need not be considered in the analysis of existing systems. The team noted in certain cases, this mass could be significant and should be considered in the piping analysis.

(2) Section 5.10.3, "Thermal Loads," states that piping systems with a maximum temperature of 150° F or less do not require a rigorous thermal analysis. The team questioned this position. The licensee stated that it was incorporating additional criteria in an upcoming revision to the specification.

(3) Section 5.10.4.1, "Seismic Inertia," contains criteria for performing seismic analyses to both the old and the new seismic criteria. In addition, the criteria allow the use of the independent support motion (ISM) method to analyze piping. However, ISM is not allowed with ASME Code Case N-411 damping and is a relatively new method of analysis. Therefore, the team requested the licensee clarify which set of design criteria provisions were used with this technique. The licensee stated that ISM had not been used at Palisades and that it would probably remove reference to it in the revision to the design specification.

(4) Section 5.10.4.1.4, "Missing Mass Correction," contains a statement that the missing mass correction does not need to be considered in the seismic analysis using the old criteria. The team noted that missing mass occurs as a result of the response spectra analysis technique which does not consider higher frequency vibrational response modes in the seismic analysis. The licensee considered this acceptable based on the assumption that the most significant portion of the mass participation would be in the flexible response modes which would be captured. However, in certain cases, such as for axial restraints on long runs of piping, the seismic loads generated in the rigid response range may be significant.

(5) Attachment G of design specification contained guidelines for reconciling the as-built piping configuration to the as-designed. These guidelines are

Electrical Power Research Institute (EPRI) Report NP-5639, "Guidelines for Piping System Reconciliation (NCIG-05, Revision 1)," May 1988 (Reference 2). The statement of scope for the document indicates the following, "the document should be restricted to piping systems analyzed and qualified on the basis of currently accepted design criteria..." The use of this document may not be consistent with some of the technical positions for criteria specified in M-195(Q) for existing designs.

Palisades Technical Specification C-173(Q) (Reference 3) details the requirements for designing and analyzing new safety-related pipe supports and for reevaluating existing or modified pipe supports. The team reviewed the referenced pipe support design specification for conformance to licensing commitments and for the consistent application of good industry practice. The team noted that the design specification for pipe supports generally addresses industry-standard design attributes for the design of new supports. However, the specification calls for the licensee to reevaluate existing or modified supports to original design criteria that are generally less stringent. The team questioned the following original design attributes that are detailed in the pipe support design specification.

(1) Section 5.4.2, "Friction Load," specifies a design friction force that is the product of the friction coefficient and the dead load rather than the product of the friction coefficient and the sum of the dead, normal, and thermal loads. This specification is unconservative since the frictional force is proportional to the magnitude of the total sustained bearing load.

(2) Section 5.4.3, "Support Self Weight," specifies that the weight of the pipe support will not be considered in the analysis and the design of the pipe support. The team noted that the effects of the weight of the support should be incorporated into the pipe support analysis as needed.

(3) Section 5.4.4, "Pipe Support Self-Weight Excitation," does not require the licensee to consider of the effects of self-weight excitation caused by seismic loads. However, this seismic load should be considered for relatively large supports. In addition, this seismic load appears to apply to structural frames such as pipe whip restraints, which were probably not originally designed to incorporate seismic load caused by self-weight excitation.

(4) Section 5.6.5, "Expansion Anchor," includes the following statement:

"Hilti Kwik Bolt Anchors installed during and after the IEBs 79-02 and 79-14 implementation shall be evaluated using the linear interaction equation of paragraph 5.6.5.d, and the anchor bolt capacities shall be in accordance with Table 5.0, except that the center-to-center spacing and the edge distance shall be 10-bolt diameters and 5-bolt diameters, respectively, regardless of embedment depth (i.e., the lower table of Table 5.0 associated with embedment depth and allowables is not applicable for existing supports)."

However, using the center-to-center spacing and edge distance specified in the paragraph instead of the values tabulated in the lower part of Table 5 appears unconservative. For example, using the 10- and 5- bolt spacing and edge criteria specified in Section 5.6.5 for a 1-1/4 inch bolt would result in a minimum required center-to-center spacing for full load capacity of 12-1/2 inches and a minimum required edge distance for full load capacity of 6-1/4 inches. However, the lower portion of Table 5 specifies that to develop the full load capacity of a 1-1/4 inch bolt embedded, 8-1/2 inches the minimum center-to-center spacing must be 17 inches and minimum edge distance must be 12-3/4 inches.

(5) Section 5.6.7, "Temperature Effect," does not require the licensee to consider thermal stresses and loads that occur within the supports for piping and tubing due to environmental temperature. However, the effect of environmental temperature should be considered for certain cases. For example, the design faulted environmental temperature specified for the containment building is 283°F. However, the industry's standard threshold for evaluating the thermal effects on piping varies from 150 to 200°F.

(6) Section 5.7.4, "Building Structure Flexibility," does not require the licensee to consider the flexibility and deflection of building steel, concrete structures, pipe whip restraints, and other building structures to which pipe supports are attached. This may not be a conservative assumption. Section 5.7.1 of the specification limits the total default deflection of the pipe support in the direction of the primary load to 1/16 inch. The licensee should evaluate the overall deflection of the support and the building structure for individual cases against this criterion to ensure that the overall deflection conforms to the piping analysis assumption that pipe supports are infinitely rigid.

(7) Section 5.9, "Expansion Anchor Bolt Force and Baseplate Analysis," discusses the use of a simplified analysis to compute anchor bolt loads for simple bolt patterns. However, the simplified method may not apply to relatively thin baseplates with relatively widely spaced bolts.

(8) Table 1, "Existing Pipe Support Design Loading Combination and Allowables," contains load combinations and allowable loads for existing pipe supports. The specification allows the normal vendor rated capacity of standard component supports, designated as catalog items in the table to be increased by 80 percent for the faulted load case. However, this increase in vendor-rated capacity is not generally specified in the vendor catalog for pipe supports designed to the requirements in the ANSI B31.1 piping code (Reference 4). Therefore, the licensee should develop a technical justification for the faulted allowables.

REFERENCES:

1. Palisades Technical Specification M-195(Q), "Requirements for the Design and Analysis of Palisades Safety Related Piping and Instrument Tubing," Revision 1, May 9, 1990.

2. EPRI Report NP-5639, "Guidelines for Piping System Reconciliation (NCIG-05), Revision 1," May 1988.

3. Palisades Technical Specification C-173(Q), "Technical Requirements for the Analysis and Design of Safety Related Pipe Supports," Revision 2, November 21, 1990.

4. American National Standard Institute Code for Pressure Piping, ANSI B31.1-1973, "Power Piping."

OBSERVATION O-2

OBSERVATION TITLE: Missing containment penetration and pipe whip restraint calculations

The licensee was not able to retrieve the original calculations for containment penetration 3 or the main steam pipe whip restraints during the period of the inspection. The team requested the original containment penetration calculation because the Reference 1 piping stress analysis for the main steam line computed large axial thermal reactions at the penetration and an adjacent pipe whip restraint due to thermal binding of the main steam line with the whip restraint. The equal and opposite axial thermal loads on the penetration and the whip restraint were due to the thermal growth of the intervening leg of piping. Containment Penetration 3 penetrates the containment liner at elevation of 616 feet. The whip restraint is supported from the containment internal structure at elevation of 623 feet 6 inches. As noted in Section 5.11.5 of the Reference 2, "Penetration Loads," the licensee was not comparing the penetration loads summarized in the Reference 1 piping stress analysis with the original containment penetration design loads because the piping loads were considered to be minor in comparison. However, this assumption may not be valid for the 100 kip axial thermal load that the Reference 1 piping stress analysis documents.

The Reference 3 civil calculation analyzed the pipe whip restraint at elevation 623 feet 6 inches for in-plane loads due to thermal binding and seismic impact. Both main steam lines bind thermally at this pipe whip restraint. The main steam line pipe whip restraints are shown on the Reference 4 drawing. The team requested the original design calculation for the pipe whip restraint to compare the analyzed piping loads due to thermal binding and seismic impact with the magnitudes of the original pipe rupture design loads.

REFERENCES:

1. Bechtel Calculation SGRP-PDS-034, "Pipe Stress Analysis of Steam Generator E50B Main Steam System Inside Containment," Revision 5, March 28, 1991.

2. Palisades Technical Specification M-195(Q), "Requirements for the Design and Analysis of Palisades Plant Safety Related Piping and Instrument Tubing," Revision No. 1, dated May 9, 1990.

3. Bechtel Calculation C-044, "Structural Adequacy Evaluation for Main Steam Line Pipe Support Loads," Revision 3, February 13, 1991.

4. Bechtel Drawing C-163, "Containment Steam Line Rupture Supports," Revision 5, March 24, 1972.

OBSERVATION 0-3

OBSERVATION TITLE: Field routed small bore safety class piping

Small bore piping 2-1/2 inches and under was originally field-routed to the generic spacing criteria specified in the Palisades FSAR (Reference 1). As noted in Subsection 2 of FSAR Section 5.7.4, "Seismic Analysis of CPCo Design Class 1 Piping":

"Piping with a fundamental natural frequency above 20 Hz was classified as rigid and analyzed statically for maximum floor accelerations. This method was generally used for small bore pipe, 2-1/2 inches and under. The rigidity requirement was achieved by limiting the piping spans to the values given in Table 5.7-5."

The team reviewed two piping stress isometrics (References 2 and 3) that depicted a 3/4-inch vent line off each of the steam generator main steam lines. The Reference 4 and 5 stress analyses use a computerized piping analysis program to explicitly qualify these lines. Each stress isometric documented a span length that exceeded the maximum span length permitted in FSAR Table 5.7-5 by about a factor of five. The Reference 2 drawing indicated a total distance between supports of about 26 feet for one horizontal and vertical leg of originally installed pipe. The Reference 3 drawing documented a similar configuration. FSAR Table 5.7-5 specified a 5 foot maximum span for 3/4-inch pipe. Although only two small bore piping isometrics were reviewed, the existing conditions caused the team to question whether small bore pipe had been consistently field-routed.

REFERENCES:

1. Consumers Company Palisades Plant Final Safety Analysis Report Amendment 14 (July 22, 1969) through Amendment 17 (Novmeber 14, 1969).

2. Bechtel Drawing M101-6141, "Steam Generator E50A M.S. Vent Pipe Removal and Reinstallation - Containment Building," Revision 1, January 22, 1991.

3. Bechtel Drawing M101-6151, "Steam Generator E50B M.S. Vent Pipe Removal and Reinstallation - Containment Building," Revision 2, March 28, 1991.

4. Bechtel Calculation No. SGRP-PDS-029, "Pipe Stress Analysis of Steam Generator E50A Main Steam Vent Line - Containment Building," Revision 5, March 28, 1991.

5. Bechtel Calculation No. SGRP-PDS-030, "Pipe Stress Analysis of Steam Generator E50B Main Steam Vent Pipe - Containment Building," Revision 5, March 28, 1991.

OBSERVATION O-4

OBSERVATION: Main steam sampling tubing material

Calculation SGRP-PDS-027 (Reference 1) and SGRP-PDS-028 (Reference 2) evaluated similar tubing routings that are connected to the main steam lines. These tubing runs are connected to short existing length of piping that were attached to the main steam line branch connections.

Both tubing calculations contained an assumption for the material properties for the existing short piece of piping attached to the main steam line branch connections. The calculations stated that this assumption did not need to be confirmed due to the short length of the existing piping and proximity to an analytical anchor. A subsequent review, during the inspection, of the piping class sheets in Specification M260 confirmed that the assumption was correct. The piping design code USAS B31.1.0-1967 (Reference 3) requirements must be met regardless of the length of piping involved. The team considered that the original disposition of the analysis assumption was technically inadequate.

REFERENCES:

1. Bechtel Calculation SGRP-PDS-027, "Stress Analysis of Main Steam Sample Tubing of Main Steam Generator E-50A Inside Containment," Revision 3, February 21, 1991.

2. Bechtel Calculation No. SGRP-PDS-028, "Stress Analysis of Main Steam Sample Tubing of Steam Generator E-50B Inside Containment," Revision 4, February 21, 1991.

3. U.S.A. Standard Code for Pressure Piping, USAS B31.1.0-1967, "Power Piping."

OBSERVATION O-5

OBSERVATION TITLE: Thermal analysis of main steam sampling tubing

Calculations SGRP-PDS-027 (Reference 1) and SGRP-PDS-028 (Reference 2) evaluated similar tubing routings that are connected to the main steam lines. The licensee performed computer analyses of the tubing for two thermal conditions. The first case considered the entire tubing at main steam temperature for steam flow through the line. The second case considered a thermal attenuation from the main steam temperature to the ambient temperature when steam is not flowing through the line.

The analysis of the second thermal case assumed that a relatively long length of tubing was at main steam temperature. The remaining tubing was assumed to be at an ambient temperature. This assumption was based on an undocumented engineering judgement. The assumption of a higher temperature in the tubing than actually exists is generally conservative since this assumption results in greater thermal expansion. However, based on the support arrangement, the team questioned if this assumption was conservative for this case. Since the tubing near the main steam line was attached to the steam generator, the assumption of main steam temperature in the tubing between the steam line and the steam generator attachment points would not result in significant differential thermal movement in that segment of tubing. The licensee agreed to further evaluate the thermal attenuation assumptions for these tubing runs.

REFERENCES:

1. Bechtel Calculation SGRP-PDS-027, "Stress Analysis of Main Steam Sample Tubing of Main Steam Generator E-50A Inside Containment," Revision 3, February 21, 1991.

2. Bechtel Calculation SGRP-PDS-028, "Stress Analysis of Main Steam Sample Tubing of Steam Generator E-50B Inside Containment," Revision 4, February 21, 1991.

3. U.S.A. Standard Code for Pressure Piping, USAS B31.1.0-1967, "Power Piping."

OBSERVATION O-6

OBSERVATION TITLE: Resolution of discrepancies found during walkdown inspections

During the walkdown inspection of the containment piping systems after the plant heatup, the licensee identified a discrepancy with steam generator B blowdown line piping support H-9. Reference 1 stated that the stress calculation SGRP-PDS-001 (Reference 2) had been revised to address the discrepancy. A similar discrepancy was identified on the steam generator A blowdown line. The discrepancy was that the blowdown piping was not bearing on the pipe support. This pipe support was the closest support to the containment penetration inside containment.

Section 8.0 of the Reference 2 calculation contained the evaluation of the discrepancy. The evaluation consisted of a statement that the additional dead weight load would be imposed on the containment penetration and that the penetration was adequate to accommodate the load increase. However, the evaluation did not address the pipe support adjacent to support H-9, which would also see an increase in deadload. The licensee stated that a further review demonstrated that the adjacent support had adequate margin to handle the increase in deadload and that the calculation would be revised to document the review.

REFERENCES:

1. Bechtel Letter BE-214, "Palisades Nuclear Plant - SGRP Bechtel Job No. 20557 Piping System Walkdown Data," March 7, 1991.

2. Bechtel Calculation SGRP-PDS-001, "Pipe Stress Analysis of Steam Generator E50B Blowdown Piping Inside Containment, Revision 9, March 28, 1991.

OBSERVATION O-7

OBSERVATION TITLE: Incomplete evaluation of torsional loads on structural members

Pipe support H-1 is a dead weight hanger on the blowdown piping for steam generator B. This hanger is supported from a horizontal length of tube steel that is welded to the bottom flange of a structural I-beam. This configuration places a local torsional load on the bottom flange of an existing structural steel tie strut. The Reference 1 calculation provided an evaluation of the hanger dead weight and the Reference 2 calculation provided an evaluation of the structural tie strut for the imposed pipe support loads.

The evaluation of the structural tie strut included an evaluation of the loads from the attached piping but the evaluation did not address the local torsional load on the bottom flange of the I-beam. Section 6.15.4 of Reference 3 states that local effects on building structural steel at pipe support attachment points shall be evaluated by the pipe support designer. The licensee subsequently evaluated the local stresses to demonstrate these stresses were within acceptable limits and agreed to revise the Reference 2 calculation to document the torsional load evaluation.

REFERENCES:

1. Becthel Calculation SGBB-PP-H1, "Pipe Support Design for Steam Generator E50B Blowdown M101-6042-H1", Revision 7, March 11, 1991.

2. Bechtel Calculation C-008, "Structural Adequacy of Interior R. C. Walls in Containment Building for S.G. E50 Blowdown System," Revision 4, August 24, 1990.

3. Palisades Technical Specification C-173(Q)," Technical Requirements for the Analysis and Design of Safety Related Pipe Supports," Revision 2, November 21, 1990.

OBSERVATION O-8

OBSERVATION TITLE: Modified blowdown and recirculation penetrations

The Reference 1 calculation qualified four modified penetrations for the design pipe rupture and piping loads. The calculation explicitly qualified blowdown penetrations 5 and 6 and qualified recirculation penetrations 16 and 55 by comparison. The pipe rupture moments and shears were applied as equivalent line loads to a finite element model of one-half of the penetration, which included the liner plate and the penetration sleeve, fin and cap plate. The piping design loads were obtained by scaling the results of the pipe rupture analysis.

The team questioned the distribution of the equivalent line load developed for the pipe rupture moment. The team also recommended that the anchor bolts be checked for a pipe rupture load combination that the calculation had not considered critical. The licensee indicated that the total pipe rupture moment that was applied to the finite element model is correct and would revise the calculation to document the method used to compute the equivalent line load. The licensee also planned to check the anchor bolts and welds for the pipe rupture load combination that combined torsional moment and concurrent shears.

REFERENCES:

Bechtel Calculation C-090, "Pipe Penetrations 5, 6, 16 & 55," Revision 2, March 13, 1991.

OBSERVATION O-9

OBSERVATION TITLE: Incomplete engineering analyses

The Reference 1 calculation specified the acceptance criteria for the performance of the auxiliary feedwater (AFW) pump. The steam generator pressure was specified as 1000 psia for calculating the required discharge pressure of the AFW pumps P-8A and P-8B. This pressure coincided with the lowest pressure of the main steam safety valves. The accumulation in the safety valve would cause the pressure in the steam generator to be higher than 1000 psia when the safety valve was relieving the mass flow being injected by the AFW pumps. Although a revision to this calculation to include the accumulation in the safety valve would show reduced available total dynamic head margins, the existing pumps still have adequate capacities to perform their safety functions.

The Reference 2 calculation documented an evaluation of the acceptability of replacing the containment sump isolation globe valves CV-1103 and CV-1104 with ball valves. The input to the analysis stated that the new valves would be "supplied to B31.1 code same as the original valves." The original valves were required to meet American Standard Association Code (ASAC) B31.1(1955) while the new ones were purchased to meet the requirements of ANSI B31.1 code (1986). The calculation neither addressed the reconciliation between the different codes nor provided references to other relevant reconciliation documents. The licensee staff stated it has already written action item record (AIR) QP-91-002 to initiate a critique of the planning, engineering, and construction of the valve replacement project because of a number concerns raised regarding activities related to the project.

The Reference 3 calculation for the valve replacement project described in the previous paragraph, dealt with piping stress analysis. This calculation did not consider the SSE loads, and the team found no statements in the calculation that justified the omission of the SSE loads in the piping stress analysis or in the pipe support loading. Also, the calculation assumed a value for the center of gravity and weight of valve DRW-151, and included the statement that the assumed data should be verified, if possible. However, the licensee could not retrieve documentation that confirmed the assumed data. This calculation appeared not to have complied with administrative procedure 9.11, "Engineering Analysis," which required that if preliminary data was used in the design, the calculation be identified as preliminary and that the data be finalized before the modification is declared operable. The licensee staff stated that the team's comments would be included as a part of the project critique initiated under AIR-QP-91-002.

The licensee revised the Reference 4 calculation to incorporate the requirement in Section 3.6.2 of the NRC Standard Review Plan that the system pressure used for pipe break analysis be commensurate with the greater of the energy contained in the system at hot standby or at 102 percent power. To comply with this requirement, the normal operating steam generator pressure was multiplied by 1.02. The licensee did not establish the steam generator pressure resulting from energy contained in the system or the system operating pressure conditions at the higher power level.

The Reference 5 calculation documented the analysis of high-energy line breaks for the AFW system. One of the items added to the calculation in Revision 1 was the statement that the break was conservatively assumed to occur during AFW injection. The calculation was based on the normal operating pressure in the steam generator of 850 psig. However, the steam generator pressure at the initiation of AFW injection would be equal to the combination of the lowest set pressure (985 psig) of the main steam safety valves and the pressure accumulated by the safety valve. The calculation failed to consider the operational conditions of the system.

The Reference 6 estimated the concrete temperatures around the penetration caused by the continuously operating blowdown system piping. The calculation specified a temperature limit of 150°F for the concrete in Attachment 1, to Reference 6 but this value was revised to 200°F in Attachment 8 without any justification. The licensee explained that paragraph A.4.1 of Reference 7 permitted a temperature of 200°F for concrete around penetrations. The team found no evidence that the licensee had reconciled the original code of construction (Reference 8) or the concrete temperature limits used in the original design with the Reference 7 code or the revised temperature limit used in the calculation.

These observations indicate that the licensee had not maintained adequate attention to detail, rigor in documenting design data and assumptions, or thoroughness in verifying the design.

REFERENCES:

1. Calculation EA-DAB-87-01, "Auxiliary Feedwater Pump Performance Requirements., January 14, 1987.

2. Calculation EA-SC-90032-01, Revision 0, "Replacement of Containment Sump Isolation Valves CV 1103 & CV 1104."

3. Calculation EA-SP-3340-PS-1, Revision 0, "Containment Sump Drains Piping - Pipe Stress Analyses Replacement Valves CV-1103 and CV-1104."

4. Bechtel Calculation SGRP-M-002, Revision 2, "Determination of Jet Impingement Forces Within Steam Generator Blowdown and Recirculation Systems."

5. Bechtel Calculation SGRP-M-010, Revision 1, "HELB Analysis for Auxiliary Feedwater System."

6. Bechtel Calculation 540-116-20557, Revision 0, "Heat Transfer Analysis of Steam Generator Blowdown Penetration."

7. American Concrete Institute Code ACI-349, "Code Requirements for Nuclear Safety Structures," 1985.

8. American Concrete Institute Code ACI-318, "Building Code Requirements for Reinforced Concrete," 1963.

OBSERVATION O-10

OBSERVATION TITLE: Updating of design drawings

The team found three instances in which the licensee had not maintained document change request (DCR) forms in the responsible engineer's file folder for drawings that had been revised, did not have a marked up copy of the revised drawing with the control copy of the drawing, and had not stamped the drawing aperture cards indicating the drawings had been revised.

Reference 1 identified the cold load values listed in the description section of the Reference 2 pipe support drawing for hanger GC4-H154 did not match the cold load values in the pipe support calculation and neither of these values was correct for the as-built condition. The licensee performed an engineering evaluation (Reference 3) to determine the correct cold and hot load settings. The licensee however had to modify the support when maintenance personnel could not adjust the support in accordance with Reference 3. The Reference 4 specification change for the support required the licensee to update two drawings (References 2 and 5). The team reviewed the control copy and aperture card for each of the drawings and found they had not been revised and had not been annotated to indicated that a revision was in progress. The engineer's file for the specification change contained no DCR forms for the required drawing changes.

The Reference 6 pipe support drawing called for a VISC-9 spring can with a 3/4 inch load bolt. Reference 7 identified that the as-built configuration included a VISC-8 spring can with a 5/8 inch load bolt. The licensee performed an engineering evaluation (Reference 8) which showed the VISC-8 spring can was acceptable. The team however found that the licensee had not completed a change to the drawing and had not annotated the control copy of the drawing or the drawing aperture card. The team found no DCR documenting the need to update the drawing.

The team reviewed the Reference 9 specification change and found similar drawing problems. Under the entry "documents to be revised," the specification change checklist included the statement "see FORM 3630 attached." The team could not locate this form in the specification change package. The specification change package contained three DCR forms that identified some of the drawings requiring revision. The team identified several other drawings that were revised as part of the change but found no DCR forms for these drawings. The team reviewed the control copies and aperture cards for these drawings and found in some cases the drawings had been updated while in other cases the drawings had not been updated and were not annotated as currently being revised.

REFERENCES:

1. Deviation report, DR-PAL-91-027.

2. Pipe support drawing, 950W18-M107, Sheet 389, Revision 3.

3. Engineering analysis, EA-D-PAL-91-027-1.

4. Specification change, SC-91-32.

5. Pipe support drawing, 950W18-M107, Sheet 399, Revision 3.

6. Pipe support drawing, 950W1-M101, Sheet 2414, Revision 4.

7. Deviation report, DR-PAL-90-107.

8. Engineering analysis, EA-SP-03356-HB-35-H933(Q).

9. Specification change, SC-90-32.

APPENDIX C

Exit Meeting Attendees

Personnel	Organization
E. G. Adensam	NRC, Acting Deputy Division Director, DRIS, NRR
J. R. Ball	NRC, Team Leader, DRIS, NRR
T. W. Bowes	CPCo, Nuclear Engineering and Construction
C. E. Brown	NRC, Project Engineer, Region III
G. D. Brown	Bechtel, Project Engineer, SGRP
C. E. Carpenter	NRC, Project Directorate III-1, NRR
D. Danielson	NRC, Section Chief, DRS, Region III
G. M. Davis	CPCo, Nuclear Engineering and Construction
E. M. Donnelly	CPCo, Director, Plant Safety and Licensing
A. Dunlop	NRC, Team Member, Region III
A. V. duBouchet	NRC Consultant
J. R. Fair	NRC Team Member, EMEB, NRR
V. Ferrarini	CPCo Consultant
R. A. Gramm	NRC, Section Chief, DRIS, NRR
J. K. Heller	NRC, Senior Resident Inspector, Palisades
D. P. Hoffman	CPCo, Vice President, Nuclear Operations
B. Holian	NRC, Palisades Project Manager, NRR
E. M. Hughes	Bechtel, Engineering Management
E. V. Imbro	NRC, Branch Chief, DRIS, NRR
R. B. Jenkins	CPCo, Nuclear Engineering and Construction
B. L. Jorgensen	NRC, Section Chief, DRP, Region III
D. C. Kansal	Bechtel, Quality Assurance Manager
E. Kubacki	CPCo, System Engineer
J. Kuemin	CPCo, Licensing Administrator
S. K. Malur	NRC, Team Member, DRIS, NRR
K. E. Marbaugh	CPCo, Quality Assurance
R. Masterson	CPCo, Consultant
H. J. Miller	NRC, Division Director, DRS, Region III
R. D. Orosz	CPCo, Nuclear Engineering and Construction Manager
T. J. Palmisano	CPCo, Administration and Planning Manager
J. C. Petro	CPCo, Quality Assurance
W. L. Roberts	CPCo, Licensing
M. A. Savage	CPCo, Public Affairs
G. B. Slade	CPCo, Plant General Manager
K. A. Toner	CPCo, Nuclear Engineering and Construction
E. Zernick	CPCo, Quality Assurance

APPENDIX D

List of Acronyms and Abbreviations

ACI	American Concrete Institute
AFW	Auxiliary Feedwater System
AIR	Action Item Record
AISC	American Institute of Steel Construction
ANSI	American National Standards Instiʼute
ASAC	American Standards Association Coue
ASME	American Society of Mechanical Engineers
CPCo	Consumers Power Company
DCC	Document Control Center
DCR	Document Change Request
DR	Deviation Report
EPRI	Electric Power Research Institute
FC	Field Change
FCR	Field Change Request
FCN	Field Change Notice
FSAR	Final Safety Analysis Report
HPSI	High Pressure Safety Injection
IEB	Inspection and Enforcement Bulletin
ISM	Independent Support Motion
KIP	Kilopound
NCR	Nonconformance Report
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
P&ID	Process and Instrumentation Diagram
PRC	Plant Review Committee
SAM	Seismic Anchor Movement
SC	Specification Change
SEP	Systematic Evaluation Program
SGRP	Steam Generator Replacement Project
SRPRP	Safety-related Piping Reverification Program
SRSS	Square Root of the Sum of the Squares
SSE	Safe Shutdown Earthquake
USAS	United States of America Standard
ZPA	Zero Period Acceleration
2-D	Two Dimensional